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Mike Skaggs
Site Vice President, Watts Bar Nuclear Plant

JUL 3 1 2006

10 CFR 50.73

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Gentlemen:

In the Matter of) Docket No. 50-390
Tennessee Valley Authority)

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1 - LICENSEE EVENT REPORT 390/2006-004 - MAIN TURBINE HIGH VIBRATION TRIP

This submittal provides LER 390/2006-004. This LER documents an event that occurred on May 30, 2006 involving a main turbine high vibration trip. The report contains information regarding this event is provided in accordance with 10 CFR 50.73 (a)(2)(iv)(A).

There are no regulatory commitments associated with this letter. Should there be questions regarding this submittal, please contact Paul L. Pace at (423) 365-1824.

Sincerely,

M Skaggs

Mike Skaggs

Enclosure
cc: See Page 2

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JUL 3 1 2006

Enclosure
cc (Enclosure):

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NRC FORM 366 (6-2004)	U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB: NO. 3150-0104	EXPIRES: 06/30/2007

LICENSEE EVENT REPORT (LER)

(See reverse for required number of digits/characters for each block)

1. FACILITY NAME	2. DOCKET NUMBER	3. PAGE
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4. TITLE
Main Turbine High Vibration Trip

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO.	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER 05000
05	30	2006	2006	- 004 -	000	07	31	2006	FACILITY NAME	DOCKET NUMBER 05000

9. OPERATING MODE: 1

10. POWER LEVEL: 100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: *(Check all that apply)*

☐ 20.2201(b)	☐ 20.2203(a)(3)(i)	☐ 50.73(a)(2)(i)(C)	☐ 50.73(a)(2)(vii)
☐ 20.2201(d)	☐ 20.2203(a)(3)(ii)	☐ 50.73(a)(2)(ii)(A)	☐ 50.73(a)(2)(viii)(A)
☐ 20.2203(a)(1)	☐ 20.2203(a)(4)	☐ 50.73(a)(2)(ii)(B)	☐ 50.73(a)(2)(viii)(B)
☐ 20.2203(a)(2)(i)	☐ 50.36(c)(1)(i)(A)	☐ 50.73(a)(2)(iii)	☐ 50.73(a)(2)(ix)(A)
☐ 20.2203(a)(2)(ii)	☐ 50.36(c)(1)(ii)(A)	☑ 50.73(a)(2)(iv)(A)	☐ 50.73(a)(2)(x)
☐ 20.2203(a)(2)(iii)	☐ 50.36(c)(2)	☐ 50.73(a)(2)(v)(A)	☐ 73.71(a)(4)
☐ 20.2203(a)(2)(iv)	☐ 50.46(a)(3)(ii)	☐ 50.73(a)(2)(v)(B)	☐ 73.71(a)(5)
☐ 20.2203(a)(2)(v)	☐ 50.73(a)(2)(i)(A)	☐ 50.73(a)(2)(v)(C)	☐ OTHER Specify in Abstract below or in NRC Form 366A
☐ 20.2203(a)(2)(vi)	☐ 50.73(a)(2)(i)(B)	☐ 50.73(a)(2)(v)(D)	

12. LICENSEE CONTACT FOR THIS LER

FACILITY NAME	TELEPHONE NUMBER *(Include Area Code)*
Rickey Stockton, Licensing Engineer	(423) 365-1818

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX
B	TA	TRB	W120	Y					

14. SUPPLEMENTAL REPORT EXPECTED	15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
○ YES *(If yes, complete 15. EXPECTED SUBMISSION DATE)* ◉ NO				

ABSTRACT *(Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)*

At 17:00:02 on May 30, 2006, main turbine vibration began increasing rapidly as indicated by a HI-HI alarm. At 17:00:16, the control room crew manually tripped the reactor and turbine when vibration reached the trip setpoint of 14 mils. Shortly after the plant trip, secondary chemistry parameters indicative of river water ingress began increasing. Following plant shutdown and cooldown, turbine and condenser inspections revealed a single blade from the last row (L-0) on the governor end of low pressure turbine (LPT) C (nearest LPT to generator) had fractured just below the rotor disk surface, causing damage to other blades on row L-0, L-0 fixed blades, surrounding turbine subcomponents, and several main condenser tubes below. Eddy current testing of all blade roots on the failed rotor identified seven more blade roots with cracks.

The root cause of the reactor trip was failure of a blade on the governor end of the LPT C rotor. The corrective actions included replacement of the LPT C rotor and repairs of collateral damage to affected main condenser tubes. TVA has also limited operation at higher backpressures. TVA also chose to increase monitoring of turbine parameters (bearings, oil quality, shaft orbits, thrust movement, etc.) during startup to ensure reliability of turbine protection subsystems.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

I. Plant Conditions:

On May 30, 2006, WBN Unit 1 was in Mode 1 at approximately 100 percent reactor thermal power. The operating temperature was 588 degrees F and Reactor Coolant System (RCS) (Energy Industry Identification System (EIIS) Code AB) pressure was 2250 psig.

II. Description of Event:

A. Event:

At 17:00:02 on May 30, 2006, main turbine (EIIS code TRB) vibration began increasing rapidly as indicated by a HI-HI alarm (EIIS code ALM). At 17:00:16, the control room crew manually tripped the reactor (EIIS code RCT) and turbine when vibration reached the trip setpoint of 14 mils. The Auxiliary Feedwater System (EIIS Code BA) automatically started as designed.

Shortly after the plant trip, secondary chemistry parameters indicative of river water ingress began increasing. Following plant shutdown and cooldown, turbine and condenser (EIIS code COND) inspections revealed a single blade from the last row (L-0) on the governor end of low pressure turbine (LPT) C (nearest LPT to generator) had fractured just below the rotor disk surface, causing damage to other blades on row L-0, L-0 fixed blades, surrounding turbine subcomponents, and several main condenser tubes below. Eddy current testing of all blade roots on the failed rotor identified seven more blade roots with cracks.

The affected rotor is a Westinghouse heavy key disc (HKD) rotor with 44" L-0 row blades (Frame # BB-281; Serial # TN-12978). The blades are 17-4 PH material, and are grouped in 4 blade groups. This rotor was installed during the first refueling outage (U1C1). The blade failure occurred after five full cycles and over 400 days of continuous operation since startup from the last refueling outage.

The actuation of the Reactor Protection (EIIS code JC) and the Auxiliary Feedwater Systems (EIIS code BA were reported in accordance with 10 CFR 50.72(b)(2)(iv) and 10 CFR 50.72(b)(3)(iv), respectively. This event is also being reported as this Licensee Event Report in accordance with 10 CFR 50.73 (a)(2)(iv).

B. Inoperable Structures, Components, or Systems that Contributed to the Event

There were no additional structures, components or systems inoperable at the start of the event that contributed to the event.

C. Dates and Approximate Times of Major Occurrences

Date	Time	Event
May 30, 2006	- 17:00:01	Unit 1 Reactor at Full Power - All conditions normal
	- 17:00:02	Turbine Vibration HI-HI Alarm
	- 17:00:16	Reactor Trip/Turbine Trip
	- 17:00:43	Generator Trip
	- 21:25	Shift personnel closed Main Steam Isolation Valves. Pressure controlled via steam generator power operated relief valves.
	- 21:30	Main condenser vacuum broken in preparation for condenser damage assessment and to reduce chemistry transient due to raw water leakage due to condenser tube damage.

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17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

D. Other Systems or Secondary Functions Affected

As a result of the turbine blade failure, a number of condenser tubes were damaged. Shortly after the plant trip, secondary chemistry parameters indicative of river water ingress began increasing.

E. Method of Discovery

As described above, this condition was first identifed when the Turbine Vibration HI-HI Alarm sounded.

F. Operator Actions

Upon receipt of the alarm, the operators manually tripped the reactor when turbine vibration setpoint was reached and began plant shutdown. Crew response to the event was timely and met Operations and Training management expectations. There were no human performance issues.

G. Safety System Responses

Upon reactor trip, Auxiliary Feedwater System started as designed.

III. CAUSE OF EVENT

The root cause of the reactor trip was failure of a blade on the governor end of the WBN Unit 1 LP Turbine rotor. As determined by metallurgical examination, the blade failed due to high cycle fatigue which preliminary analyses indicates was the result of vibration conditions due to operation at the high end of approved condenser back pressure limits.

IV. ANALYSIS OF THE EVENT

Plant safety systems functioned normally in response to the reactor trip. All rods inserted fully. Steam generator level was maintained initially via the Auxiliary Feedwater system. However, several secondary side components failed to operate as required.

When the L-0 blade on LP-C turbine failed it was ejected into the C Zone main condenser and damaged several tubes allowing raw river water to enter the main condenser. Due to the additional flow into the main condenser the condensate storage tank overflowed due to back flow from the hotwell. Cooldown of the plant continued utilizing the AFW pumps and main condenser for approximately 4.5 hours. In order to facilitate repairs, cooling was shifted to the S/G PORVs to allow main condenser inspection. Subsequently, Residual Heat Removal (RHR) (EIIS code BP) cooling was placed into service.

Because of the damaged tubes in the main condenser and the need to use AFW from the condensate storage tank (CST) (EIIS code KA/TK) the secondary side including the steam generators became contaminated with raw water. Once the plant was placed on RHR cooling, the use of AFW was no longer needed.

V. ASSESSMENT OF SAFETY CONSEQUENCES

This manual reactor trip can be compared to the Final Safety Analysis Report (FSAR) Loss of External Electrical Load and/or Turbine Trip in UFSAR section 15.2.7 (page 15.2-21). The manual trip occurred as a result of observed high turbine vibrations. The vibrations were indicated by instrumentation and felt by control room personnel. Turbine vibrations were the result of an unbalanced main turbine rotor following catastrophic LP-C blade failure. The blade also damaged tubes in the main condenser in both the east and west waterbox. The damaged condenser tubes then provided a raw water path to the hotwell. Rising hotwell water level caused a diversion to the CST and, upon initiation of auxiliary feedwater, a path for raw water ingress into

17. NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)*

the steam generators. The plant was stabilized using steam dumps. Eventual cooldown was through transition to steam generator atmospheric relief valves (ADVs) (EIIS code V) and later to RHR. The reactor coolant system responded to the initial transient as expected with no pressurizer power operated relief valve (PORV) actuation, no safety injection (EIIS code BP) initiation, and no steam generator ADV actuation.

The FSAR 15.2.7 analysis contains several analysis conservatisms which were not characteristic of the actual event. The FSAR analysis assumes the reactor trip is based on a reactor protection system trip setpoint exceedance rather than a direct manual trip. In addition, reactor control is assumed to be in manual, no credit is taken for the steam dump system, and no credit is taken for the steam generator atmospheric relief valves (only steam generator safeties are credited). The actual event had automatic rod control and steam dumps available. The FSAR analysis demonstrates for two cases (DNB case where credit is taken for the pressurizer PORVs and spray, and RCS overpressure case where no credit is taken for the pressurizer spray or PORVs) that the minimum DNBR is well above the limiting value and that the RCS pressure safety analysis limits are met.

Therefore based upon the above, the actual event is bounded by the FSAR safety analysis assumptions.

V. CORRECTIVE ACTIONS

A. Immediate Corrective Actions

Operators responded to the plant transient in accordance with appropriate plant procedures. An event team was assembled to investigate the cause of the event.

TVA has replaced the damaged rotor with a spare rotor. The condenser and hotwell were inspected, tube plugged or damaged sections removed. Secondary side was drained and flushed to clean raw water contaminants from the systems. Metallugical analysis of failed blades was performed to support the root cause analysis.

B. Corrective Actions to Prevent Recurrence (TVA does not consider these items to constitute regulatory commitments. TVA's corrective action program tracks completion of these actions.)

Backpressure limits were lowered and associated procedures changed to reduce possibility of future events due to high cycle fatigue.

VI. ADDITIONAL INFORMATION

A. Failed Components

The affected rotor is a Westinghouse heavy key disc (HKD) rotor with 44" L-0 row blades (Frame # BB-281; Serial # TN-12978). The blades are 17-4 PH material, and are grouped in 4 blade groups. This rotor was installed during the first refueling outage (U1C1). The blade failure occurred after five full cycles and over 400 days of continuous operation since startup from the last refueling outage.

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B. Previous LERs on Similar Events

A review was performed of the previous WBN Licensee Event Reports (LERs) for any events associated with turbine high vibration exceeding procedure limits which resulted in a reactor trip. There were no previous LERs identified which were associated with high vibration of the turbine generator which caused a reactor trip.

C. Additional Information:

None.

D. Safety System Functional Failure

This event did not involve a safety system functional failure as defined in NEI 99 02, Revision 4.

E. Loss of Normal Heat Removal Consideration

This event is not considered a scram with loss of normal heat removal.

VII. COMMITMENTS

None